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06/12/98
Rev. 06

ENGINEERING DESIGN FILE

Functional File No. 1000-137
EDF No. TRA-2000-004
Page 1 of 1

1. Project File No. N/A 2. Project/Task TRA FACILITY HAZARDS CHARACTERIZATION
3. Subtask TRA FACILITIES HAZARD CHARACTERIZATIONS

4. Title: TRA-704, -705, -706, -755, AND TRA-642 REACTOR VESSEL RADIOLOGICAL SOURCE TERM DETERMINATIONS

5. Summary:

DOE Order 5480.23, *Nuclear Safety Analysis Reports (SAR)*, requires that contractors perform a safety analysis that develops and evaluates the adequacy of the safety basis for each facility. A SAR or other applicable safety document is prepared by the contractor, which documents the safety analysis. A facility assessment that aids in determining the hazardous classification is an integral part of this analysis. The hazardous classification process requires identification of all hazards including radioactive materials. This Engineering Design File (EDF) documents the radiological source term for various buildings/facilities at the Test Reactor Area (TRA).

This EDF is intended to serve as a coversheet that will ensure a permanent record is available for TRA radiological source term determinations for buildings/facilities that have been addressed using a NOTEGRAM versus a separate EDF. The details are included in the following attachments:

1. TRA-706 ETR DELAY TANKS RADIOLOGICAL SOURCE TERM DATA-(DEH-11-00)
2. TRA-642 ETR REACTOR VESSEL RADIOLOGICAL SOURCE TERM DATA- (DEH-12-00)
3. TRA-704 AND TRA-705 ETR PRIMARY AND SECONDARY FILTERS RADIOLOGICAL SOURCE TERM DATA-(DEH-13-00)
4. TRA-755 ETR LOOP FILTERS RADIOLOGICAL SOURCE TERM DATA-(DEH-14-00)

6. Distribution (complete package): BBWI Radiological Control Files, MS 4138 (Original); TRA Radiological Control Files, MS 7110; J. D. Edelmayer, MS 7110; TRA Landlord Files, MS 7121; G. A. Dinneen, MS 3870

Distribution (summary package only): C. D. Morgan, MS 7110, C. E. Mills, MS 7123

EDF Summary				
Author	R	D. E. Movis	<i>[Signature]</i>	5-4-00
Independent Verification	R	B. L. Grant	<i>[Signature]</i>	5-9-00
Requestor	R	J. D. Edelmayer	<i>[Signature]</i>	5/10/00
INEEL Radiological Control	A	A. N. Singer	<i>[Signature]</i>	5/31/00
Director			<i>[Signature]</i>	5/31/00

NOTEGRAM

TO: G. Dinneen, MS #3870
FROM: D. Hovis, MS #7110 *Darin Hovis*
DATE: April 27, 2000
SUBJECT: TRA-706 ETR DELAY TANKS RADIOLOGICAL SOURCE TERM DATA - (DEH-11-00)

DOE Order 5480.23, *Nuclear Safety Analysis Reports (SAR)*, requires that contractors perform a safety analysis that develops and evaluates the adequacy of the safety basis for each facility. A SAR or other applicable safety document is prepared by the contractor, which documents the safety analysis. A facility assessment that aids in determining the hazardous classification is an integral part of this analysis. The hazardous classification process requires identification of all hazards including radioactive materials. This documents the radiological source term typically introduced to Structure TRA-706 at the TRA.

The results of the assessment indicate that no calculable source term exists for the delay tanks.

FACILITY BACKGROUND AND CURRENT USAGE:

Prior to deactivation, the delay tanks were in the gaseous effluent processing line for the On-Line Cover Gas System. Normal flow from the OLCS system was directly to the delay tanks. Depending on the rate of exhaust flow, the delay tanks could delay the flow discharged to the exhaust stack for up to 60 days. From the delay tanks the effluent discharged either to the secondary filters or to a 3-hp exhaust fan. The tanks are located underground east of the compressor building inside a fenced Soil Contamination Area measuring approximately 100 ft. x 180 ft.

RADIOLOGICAL SOURCE TERM:

Radiological surveys dated 9/14/99 indicate <0.1 mr/hr general area and contact radiation levels throughout the delay tanks. Numerous smears taken around the tanks indicate no detectable loose surface contamination. Since the tanks have been deactivated for quite a few years and given this survey data, there is no measurable radiological source term. The delay tanks are assumed to be internally contaminated. Should areas of the facility that were previously unavailable for radiological surveys be made accessible, further sampling and surveys may more accurately identify radiological constituents, if any are present.

ATTACHMENTS:

1. Radiological Survey Report dated 9/14/99.

deh *deh*

cc: J. D. Edelmayer, MS 7110 *J. D. Edelmayer*
TRA Landlord Files, MS 7121
TRA Radiological Control Files, MS 7110

441.45#
10/10/97
Rev. #03

RADIOLOGICAL SURVEY REPORT

BARCODE ORIGINAL

BLDG.: Delay Tanks
AREA/ROOM: TRA
RWP #: 9900067-104
LOG #: 3
DATE: 9/19/99
TIME: 11:30

☐ ROUTINE

JOB DESCRIPTION

☒ NON ROUTINE (SPECIFY) ☐ FOLLOW UP

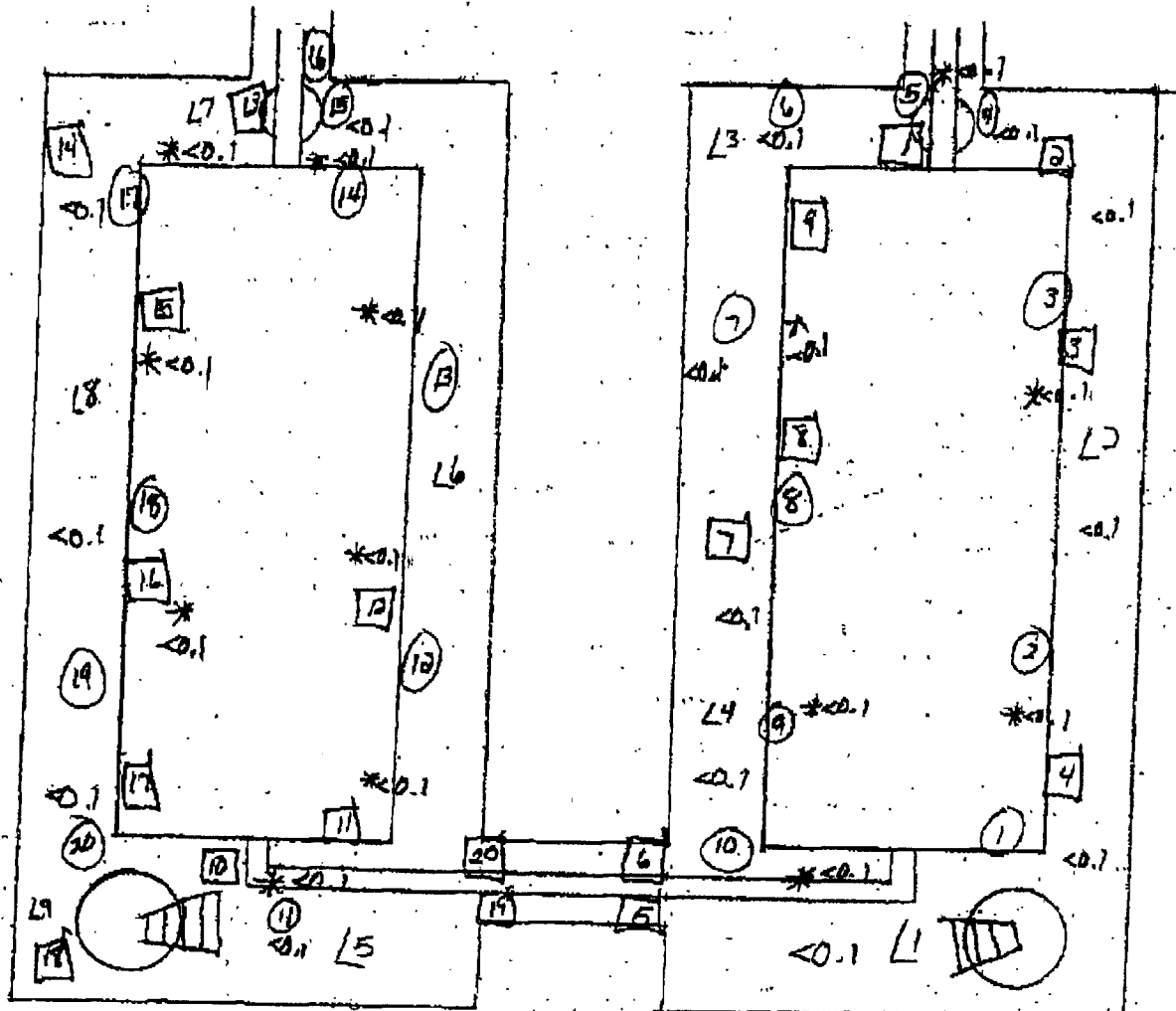
COMMENTS:

Point to Sample

RCT:

PRINT/SIGNATURE

REVIEWED BY:



NOTEGRAM

TO: G. Dinneen, MS #3870
FROM: D. Hovis, MS #7110 *Darrin E. Hovis*
DATE: April 27, 2000
SUBJECT: **TRA-642 ETR REACTOR VESSEL RADIOLOGICAL SOURCE TERM DATA - (DEH-12-00)**

DOE Order 5480.23, *Nuclear Safety Analysis Reports (SAR)*, requires that contractors perform a safety analysis that develops and evaluates the adequacy of the safety basis for each facility. A SAR or other applicable safety document is prepared by the contractor, which documents the safety analysis. A facility assessment that aids in determining the hazardous classification is an integral part of this analysis. The hazardous classification process requires identification of all hazards including radioactive materials. This documents the radiological source term remaining in the ETR reactor vessel in Building TRA-642 at the TRA.

The results of the assessment indicate the following radiological source term exists for the reactor vessel:

Co-60 =	2850 Curies
Cs-137 =	402 Curies
Sr-90 =	3.08 Curies

FACILITY BACKGROUND AND CURRENT USAGE:

The pressure vessel has been completely drained with its internal equipment (core filler pieces, reflectors, thermal shield, etc., excluding fuel) in place. The fuel was removed in the 1982 timeframe. Many spare non-fuel irradiated pieces (grid adapters, x-baskets, c-4x pieces, and R-4x pieces) are stored on top of the core area; poison sections, shock sections, and spacers are stored in the control rod guide tubes. The reactor vessel top dome and removable biological shielding are installed in place.

RADIOLOGICAL SOURCE TERM:

Information obtained from the *Characterization of the Engineering Test Reactor Facility*, EGG-PR-5784, September 1982, indicates that 98% of the radioactivity contained in the core area is due to Co-60 and was calculated to be 30,400 curies. The other 2% of the activity would be comprised of mixed fission products and activation/corrosion products. This 2% would amount to 608 curies of activity. Information obtained in the *Safety Analysis Report for the Engineering Test Reactor Facility in an Inactive Status*, July 1991, describes in-situ gamma-ray spectrometry measurements at several core positions that revealed the predominant radioactive species to be Co-60 with traces of Cs-137, Fe-59, Mn-54, and Zr-95. The corrosion and activation products would have long since decayed to a negligible amount. Therefore the only nuclides which would still be present in any meaningful quantity would be Co-60 and Cs-137, with Cs-137 representing the majority of the 2% of the activity not attributed to Co-60. Assuming 18 years since the calculation of the activity levels, the activities of the radionuclides were calculated by using the following equation:

$$A_f = A_0 e^{-\lambda t}$$


G. Dinneen, MS #3870
NOTEGRAM - DEH-12-00
April 27, 2000
Page 2 of 2

Strontium activity was then determined by scaling to Cs-137 using a scaling factor of $7.66E-03$ found in EDF-TRA-007, *Engineering Test Reactor Radiological Characterization*, dated June 27, 1996.

ATTACHMENTS:

1. Page 209 of *Characterization of the Engineering Test Reactor Facility*, EGG-PR-5784, September 1982.
2. Page 6-46 of *Safety Analysis Report for the Engineering Test Reactor Facility in an Inactive Status*, July 1991.

dch 

cc: J. D. Edelmayer, MS 7110 
TRA Landlord Files, MS 7121
TRA Radiological Control Files, MS 7110

4.2.1 Reactor Pressure Vessel

Routine radiological data collecting methods could not be used to survey the reactor pressure vessel after it was drained because of increased radiation levels. The increase in radiation required that the biological shielding be in place prior to draining the vessel, thus precluding internal radiological measurements. Therefore other means were employed to radiologically characterize this component. In-vessel radiological data were taken before draining the vessel. Radiological data outside the biological shield were taken during a trial draindown and during the final draining of the vessel.

The core area and grid plate radiological data were taken with the water level in the vessel at approximately 17 ft above the core area. A dry tube was placed in nine different positions within the core area (see Figure 61) and a Victoreen Model 510 roentgen rate meter was lowered into the tube for radiation measurements. It was assumed that the bottom of the tube was located at the top surface of the reactor core grid plate. Readings were taken at four elevations within the dry tubes (see Table 1). The higher readings above the grid plate in the E-10 core position are assumed to be the result of high radiation from the stainless steel F-10 in-pile tube which had been exposed to the entire fuel element length flux during previous power operation. The cobalt-60 activity of the grid plate was calculated to be 3.04×10^4 Ci, which is estimated to be 98% of all the activity in the core area components, reflector, and vessel components. Measured radiation levels support this calculated value. Using a gold foil dosimeter lowered down a dry tube in the water filled vessel, a thermal neutron flux of $900 \text{ n/cm}^2\text{s}$ was measured on top of the beryllium. After the reactor vessel was drained, however, scanning detected no neutron activity external to the biological shielding.

requirement per Section 4.2 is actually shown under inactivation of the canal facilities (see Section 6.2.24).

2. Special Nuclear Materials: Removal of the two neutron sources eliminated this hazard. Removal from the facility satisfies the SFMP special nuclear materials removal requirement (see Section 4.2).
3. Radiological: Proximity and long exposure to the fuel irradiation caused considerable activation effects. During a partial vessel drain test, gamma-ray spectrometry measurements at several core positions showed the radioactive species to be predominately Co-60 with traces of Cs-137, Fe-59, Mn-54 and Zr-95 (Reference 99). Major contributors of the cobalt are the grid plate and the water loop in-pile tubes.

Prior to the preliminary PCS draining, thermal and radiological analyses were performed to predict results of removing the coolant. Grid plate and inner tank temperature due to the decay heat were predicted to be less than 250°F after 36 hours of being drained (Reference 100). Radiological analysis predicted beta/gamma radiation above the removable biological shielding to 1.7×10^{-6} R/hr upon being drained (Reference 14). Results of the preliminary drain showed the maximum component temperature to be 80°F after 48 hours of being drained (Reference 101 reported 105°F but a 25°F high offset error was subsequently found lowering the test results). Also, there was no detectable change in the beta/gamma radiation fields outside the biological shielding before and after the draining. No fast neutron radiation was detected before or after the draining. The higher temperature predicted was partially due to a decay heat source term several times larger than later determined for the final radiological analysis in Reference 14. Local radiation field effects around the biological shield on the main floor were found to mask any effect expected from the vessel draining.

NOTEGRAM

TO: G. Dinneen, MS #3870
FROM: D. Hovis, MS #7110 *Dustin E. Hovis*
DATE: May 3, 2000
SUBJECT: TRA-755 ETR LOOP FILTERS RADIOLOGICAL SOURCE TERM
DATA - (DEH-14-00)

DOE Order 5480.23, *Nuclear Safety Analysis Reports (SAR)*, requires that contractors perform a safety analysis that develops and evaluates the adequacy of the safety basis for each facility. A SAR or other applicable safety document is prepared by the contractor, which documents the safety analysis. A facility assessment that aids in determining the hazardous classification is an integral part of this analysis. The hazardous classification process requires identification of all hazards including radioactive materials. This documents the radiological source term of the loop filters in TRA-755 at the TRA. The results of the assessment indicate the following total source term for the filters:

Cobalt-60 = 6.9 Ci Barium-137m = 131.9 Ci Cesium-137 = 138 Ci Strontium-90 = 1.06 Ci

FACILITY BACKGROUND AND CURRENT USAGE:

The hot filter pit (TRA-755) houses three filters, designated the Loop 33, Loop 66, and Loop 99 hot filters. These filters are a canister activated-charcoal type contained in a lead and concrete shield. The filter canisters themselves are constructed with steel. Filters 66 and 99 are identical in size and shape. Filter 33 is smaller than the other two.

RADIOLOGICAL SOURCE TERM:

A conservative approach to estimating the radioactivity in the loop filters was followed. All filters are assumed to contain the same levels of radioactivity. Radiological survey data reported in the *Characterization of the Engineering Test Reactor Facility*, EGG-PR-5784, September, 1982, showed contact radiation levels on one of the filters of 650 mr/hr. This value was used in MicroShield v5.05 to model the activity of the filter. Allowing for decay since 1982, Co-60 activity would be approximately 3 curies. Strontium and cesium activities were then determined by scaling to Co-60 using scaling factors and ratios found in EDF-TRA-007, *Engineering Test Reactor Radiological Characterization*, dated June 27, 1996. The resultant activity was then multiplied by two to account for both of the large filters. The same dose rate (650 mr/hr) was then used with the smaller filter to obtain its activity and the same process was used to decay and estimate the activities of the other radionuclides. Finally, the activities of all three filters were added together to estimate the total source term.

ATTACHMENTS:

1. Activity decay calculation sheet.
2. MicroShield v5.05 Calculation Sheets; DOS File #6699FILT.MS5, and 33FILT.MS5.

deh *deh*

cc: J. D. Edelmayer, MS 7110 *J. D. Edelmayer*
TRA Landlord Files, MS 7121
TRA Radiological Control Files, MS 7110

$$A_f = A_o e^{-\lambda t}$$

where: A_f = Final Activity

A_o = Initial Activity

λ = Decay Constant = $\frac{0.693}{t \frac{1}{2}}$

t = Time

Assume that the measured radiation is due to Co-60 activity in the filter.

For Filters #66 and #99:

$$A_f = (32 \text{ Ci}) e^{-(0.693/5.27 \text{ y})(18 \text{ y})}$$

$$A_f = 3 \text{ Ci}$$

For Filter #33:

$$A_f = (9.45 \text{ Ci}) e^{-(0.693/5.27 \text{ y})(18 \text{ y})}$$

$$A_f = 0.9 \text{ Ci}$$

MicroShield v5.05 (5.05-00086)
Bechtel Idaho

Page : 1
 DOS File: 6699FILT.MS5
 Run Date: May 17, 2000
 Run Time: 8:30:09 AM
 Duration: 00:00:12

File Ref: DEH-14-
 Date: 5-17-0
 By: Darrin P.
 Checked: 5-17-0

Case Title: Filters
Description: ETR LOOP FILTERS #66 AND #99
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions
 Height 243.84 cm 8 ft
 Radius 68.58 cm 2 ft 3.0 in

Dose Points

	X	Y	Z
# 1	78.74 cm	121.92 cm	0 cm
	2 ft 7.0 in	4 ft	0.0 in

Shields

Shield Name	Dimension	Material	Densit
Source	2.20e+05 in ³	Carbon	2.25
Transition		Air	0.0012
Air Gap		Air	0.0012
Wall Clad	3.0 in	Iron	7.86
Top Clad	2.0 in	Iron	7.86

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	$\mu\text{Ci/cm}^3$	Bq/cm ³
Ba-137m	6.0544e+002	2.2401e+013	1.6804e+002	6.2176e+006
Co-60	3.2000e+001	1.1840e+012	8.8818e+000	3.2863e+005
Cs-137	6.4000e+002	2.3680e+013	1.7764e+002	6.5725e+006

Buildup
 The material reference is : Wall Clad

Integration Parameters

Radial	20
Circumferential	20
Y Direction (axial)	20

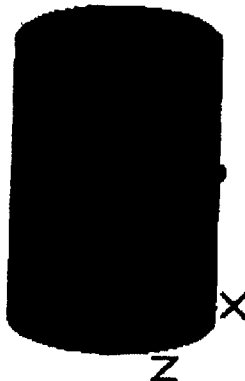
Energy MeV	Activity photons/sec	Fluence Rate		Exposure Rate	
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.0318	4.638e+11	4.156e-175	1.463e-21	3.462e-177	1.218e-23
0.0322	8.557e+11	5.678e-169	2.742e-21	4.570e-171	2.207e-23
0.0364	3.114e+11	1.036e-118	1.202e-21	5.886e-121	6.828e-24
0.6616	2.016e+13	2.014e+04	1.961e+05	3.905e+01	3.801e+02
0.6938	1.931e+08	2.315e-01	2.186e+00	4.469e-04	4.220e-03
1.1732	1.184e+12	9.763e+03	6.256e+04	1.745e+01	1.118e+02
1.3325	1.184e+12	1.517e+04	8.829e+04	2.632e+01	1.532e+02
TOTALS:	2.416e+13	4.508e+04	3.469e+05	8.282e+01	6.450e+02

MicroShield v5.05 (5.05-00086)
Bechtel Idaho

Page : 1
 DOS File: 33FILT.MS5
 Run Date: May 16, 2000
 Run Time: 4:43:50 PM
 Duration: 00:00:10

File Ref: DEH-14
 Date: 5-17-00
 By: Carrin
 Checked: 7/2/00

Case Title: Filters
Description: ETR LOOP FILTER #33
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions
 Height 152.4 cm 5 ft 0.0 in
 Radius 45.72 cm 1 ft 6.0 in

Dose Points

	X	Y	Z
# 1	55.88 cm	76.2 cm	0 cm
	1 ft 10.0 in	2 ft 6.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.00e+06 cm ³	Carbon	2.25
Transition		Air	0.0012
Air Gap		Air	0.0012
Wall Clad	7.62 cm	Iron	7.86
Top Clad	5.08 cm	Iron	7.86

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	$\mu\text{Ci}/\text{cm}^3$	Bq/cm ³
Ba-137m	1.7879e+002	6.6154e+012	1.7865e+002	6.6101e+006
Co-60	9.4500e+000	3.4965e+011	9.4424e+000	3.4937e+005
Cs-137	1.8900e+002	6.9930e+012	1.8885e+002	6.9874e+006

Buildup
 The material reference is : Wall Clad

Integration Parameters

Radial	20
Circumferential	20
Y Direction (axial)	20

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm ² /sec No Buildup	Results		Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
			Fluence Rate MeV/cm ² /sec With Buildup			
0.0318	1.370e+11	3.470e-174	8.960e-22		2.890e-176	7.464e-24
0.0322	2.527e+11	4.446e-168	1.680e-21		3.578e-170	1.352e-23
0.0364	9.195e+10	4.330e-118	7.363e-22		2.460e-120	4.183e-24
0.6616	5.953e+12	2.067e+04	1.997e+05		4.007e+01	3.872e+02
0.6938	5.703e+07	2.373e-01	2.226e+00		4.582e-04	4.297e-03
1.1732	3.497e+11	9.938e+03	6.340e+04		1.776e+01	1.133e+02
1.3325	3.497e+11	1.542e+04	8.939e+04		2.676e+01	1.551e+02
TOTALS:	7.133e+12	4.603e+04	3.525e+05		8.459e+01	6.556e+02

NOTEGRAM

TO: G. Dinneen, MS #3870
FROM: D. Hovis, MS #7110
DATE: May 3, 2000
SUBJECT: TRA-704 AND TRA-705 ETR PRIMARY AND SECONDARY FILTERS
RADIOLOGICAL SOURCE TERM DATA - (DEH-13-00)

DOE Order 5480.23, *Nuclear Safety Analysis Reports (SAR)*, requires that contractors perform a safety analysis that develops and evaluates the adequacy of the safety basis for each facility. A SAR or other applicable safety document is prepared by the contractor, which documents the safety analysis. A facility assessment that aids in determining the hazardous classification is an integral part of this analysis. The hazardous classification process requires identification of all hazards including radioactive materials. This documents the radiological source term of the primary and secondary filters in TRA-704 and TRA-705 at the TRA.

The results of the assessment indicate the following total source term for both sets of filters:

Cobalt-60 = 8.4 Ci
Cesium-137 = 168 Ci
Barium-137m = 159.6 Ci
Strontium-90 = 1.29 Ci

FACILITY BACKGROUND AND CURRENT USAGE:

Exhaust piping is routed from the reactor building to the waste gas stack. This routing is provided in underground tunnels with pits having filters and delay tanks. The primary filter pit (TRA-704) houses two filters and is 13 ft. 8 in. long, 9 ft. 6 in. wide, and about 17 ft. deep. The pit can be entered through hatches which require a crane to lift them. The secondary filter pit (TRA-705) also houses two filters and is the same size as the primary filter pit. All four of the filters are of the same construction and dimensions. The filter elements are glass wool and absolute filters. The filters are shielded by 4 ¾ inches of lead all around the sides, and have steel, lead, and concrete shielding on top. The filters were deactivated when the reactor was shut down in the 1981-1982 time frame.

RADIOLOGICAL SOURCE TERM:


A conservative approach to estimating the radioactivity in the primary and secondary filters was followed. All filters are assumed to contain the same levels of radioactivity since the normal flowpath of the effluent gases flowed through both sets of filters. Radiological survey data reported in the *Characterization of the Engineering Test Reactor Facility*, EGG-PR-5784, September, 1982, showed contact radiation levels on one of the primary filters of 10 mr/hr. This value was used in MicroShield v5.05 to model the activity of the filter. Allowing for decay since 1982, Co-60 activity would be approximately 2.1 curies. Strontium and cesium activities were then determined by scaling to Co-60 using scaling factors and ratios found in EDF-TRA-007, *Engineering Test Reactor Radiological Characterization*, dated June 27, 1996. The resultant activity was then multiplied by four to account for all the filters.

G. Dinneen, MS #3870
NOTEGRAM - DEH-13-00
May 3, 2000
Page 2 of 2

ATTACHMENTS:

1. Activity decay calculation sheet.
2. MicroShield v5.05 Calculation Sheet; DOS File #FILTERS.MS5

deh 

cc: J. D. Edelmayer, MS 7110 
TRA Landlord Files, MS 7121
TRA Radiological Control Files, MS 7110

$$A_f = A_o e^{-\lambda t}$$

where: A_f = Final Activity

A_o = Initial Activity

λ = Decay Constant = $\frac{0.693}{t \frac{1}{2}}$

t = Time

Assume that the measured radiation is due to Co-60 activity in the filter.

$$A_f = (22.5 \text{ Ci}) e^{-(0.693/5.27 \text{ y})(18 \text{ y})}$$

$$A_f = 2.1 \text{ Ci}$$

MicroShield v5.05 (5.05-00086)
Bechtel Idaho

Page : 1
 DOS File: FILTERS.MS5
 Run Date: May 17, 2000
 Run Time: 8:03:58 AM
 Duration: 00:00:12

File Ref: DEH-13-
 Date: 5-17-00
 By: *Chris E*
 Checked: *ESL*

Case Title: Filters
Description: ETR primary and secondary filters
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions
 Height 172.085 cm 5 ft 7.8 in
 Radius 73.66 cm 2 ft 5.0 in

Dose Points

#	X	Y	Z
1	88.265 cm 2 ft 10.8 in	76.2 cm 2 ft 6.0 in	0 cm 0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.79e+05 in ³	Glass Wool	0.064
Transition		Air	0.0012
Air Gap		Air	0.0012
Wall Clad	4.75 in	Lead	11.3
Top Clad	2.0 in	Iron	7.86

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	$\mu\text{Ci/cm}^3$	Bq/cm ³
Ba-137m	4.2570e+002	1.5751e+013	1.4513e+002	5.3697e+006
Co-60	2.2500e+001	8.3250e+011	7.6705e+000	2.8381e+005
Cs-137	4.5000e+002	1.6650e+013	1.5341e+002	5.6762e+006

Buildup
 The material reference is : Wall Clad

Integration Parameters

Radial	20
Circumferential	20
Y Direction (axial)	20

Energy MeV	Activity photons/sec	Results			
		Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.0318	3.261e+11	0.000e+00	9.705e-22	0.000e+00	8.084e-24
0.0322	6.016e+11	0.000e+00	1.814e-21	0.000e+00	1.460e-23
0.0364	2.189e+11	0.000e+00	7.551e-22	0.000e+00	4.290e-24
0.6616	1.417e+13	5.998e+00	1.848e+01	1.163e-02	3.582e-02
0.6938	1.358e+08	1.366e-04	4.305e-04	2.637e-07	8.312e-07
1.1732	8.325e+11	4.598e+02	1.695e+03	8.217e-01	3.029e+00
1.3325	8.325e+11	1.156e+03	4.305e+03	2.006e+00	7.470e+00
TOTALS:	1.698e+13	1.622e+03	6.019e+03	2.840e+00	1.053e+01

